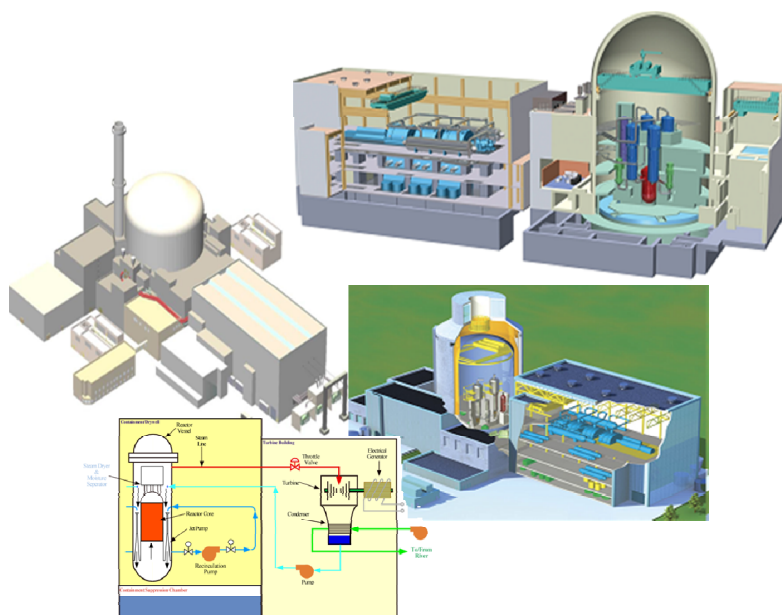




NUCLEAR REGULATORY COMMISSION

Reactor Technology Training Branch



Part I

Introduction to Reactor Technology - PWR

Chapter 1.0 Introduction to Pressurized Water Reactor Systems

UNITED STATES
NUCLEAR REGULATORY COMMISSION
HUMAN RESOURCES TRAINING & DEVELOPMENT

Introduction to Reactor Technology

This manual is a text and reference document for the Introduction to Reactor Technology. It should be used by students as a study guide during attendance at this course. This manual was compiled by staff members from the Human Resources Training & Development in the Office of Human Resources.

The information in this manual was compiled for NRC personnel in support of internal training and qualification programs. No assumptions should be made as to its applicability for any other purpose. Information or statements contained in this manual should not be interpreted as setting official policy. The data provided are not necessarily specific to any particular nuclear power plant, but can be considered to be representative of the vendor design.

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The information contained in this chapter pertains to current operational reactor designs. Advanced reactor designs are provided in separate chapters.

1.0 INTRODUCTION TO PWR SYSTEMS

1.0.1 Introduction

The pressurized water reactor (PWR) is one of the two light-water reactor designs used in the United States for the generation of electricity. This section will discuss the general layout of a PWR, point out some of the major mechanical systems and their location, and discuss some of the control systems used at the plant. In order to discuss these topics in this and later chapters, Figure 1.0-1 shows some of the symbols that will be used in the diagrams. These are typical symbols used in most mechanical and electrical drawings.

1.0.2 General Description

A pressurized water reactor (PWR) generating system is a dual cycle plant consisting of a closed, pressurized, reactor coolant system (primary) and a separate power conversion system (secondary) for the generation of electricity. The use of a dual cycle keeps the potentially radioactive reactor coolant separate from the main turbine, condenser, and other secondary plant components.

1.0.2.1 Primary Cycle

The composite flow diagram shown in Figure 1.0-2 illustrates the dual cycle nature of a PWR. The primary cycle, or reactor coolant system (RCS), starts at the reactor, where the heat from fission is transferred to the light water coolant. The hot reactor coolant flows to the steam generators via the reactor coolant system hot legs. In the steam generators, the hot reactor coolant flows inside the steam generator u-tubes and gives up its energy to the secondary coolant on the outside of the tubes. The slightly cooler reactor coolant then travels to the suction of the reactor coolant pumps via the intermediate legs. The reactor coolant pumps increase the pressure of the coolant and returns it to the reactor via the reactor coolant system cold legs. The rated thermal output of a PWR is determined by the size of the reactor core and the number of heat transfer loops in the primary system.

1.0.2.2 Secondary Cycle

The secondary, or steam cycle, begins in the shell side of the steam generators, where the incoming feedwater is boiled as it picks up heat from contacting the U-tubes containing hot reactor coolant. The steam generator U-tubes provide the barrier between the primary and secondary cycles.

Steam leaving the steam generators passes through steamline isolation valves and is directed to the high pressure section of the main turbine. After leaving the high pressure turbine, the low energy, moisture laden steam is routed to a moisture separator/reheater,

where the excess moisture is removed and a small amount of superheat is applied by reheating the steam with high energy steam from the main steam system.

The dry, reheated steam then enters the low pressure turbines, where most of its remaining available energy is removed, and exits to the main condenser. Provisions are made to bypass the turbine and dump steam directly to the main condenser under certain plant conditions.

In the condenser, the steam is condensed by passing over tubes containing relatively cold condenser circulating water and is collected in the condenser hotwell. Condensate pumps take a suction on the hotwell and pump the water through the tube side of feedwater heaters to the suction of the main feedwater pumps.

The feedwater heaters are provided to increase plant efficiency. The main feedwater pumps discharge, through level control valves, into the steam generators, where the feedwater is boiled to produce steam, and the cycle begins again.

1.0.2.3 Support and Emergency Systems

The component cooling water (CCW) system provides a cooling medium to various potentially radioactive components, such as heat exchangers, pump oil and seal coolers, and fan units. It is a closed loop system and is cooled by the service water system. The service water system will transfer the heat to the environment by directly taking a suction on and discharging to a lake, ocean, or river, or via a cooling tower.

The chemical and volume control system (CVCS) maintains the purity of the RCS by means of demineralizer beds that continuously purify a small letdown stream from the RCS. This purified water is returned to the RCS at a rate which is controlled to maintain the proper pressurizer level. A portion of the CVCS serves as a high pressure supply of borated water to the RCS in emergency situations.

The residual heat removal (RHR) system, located in the auxiliary building, serves two functions. The normal function is to remove the decay heat from the core after shutdown. This is accomplished by pumping the hot RCS water from the hot leg through a heat exchanger and back into the RCS via the cold leg. The accident function is to pump cool, borated water from the refueling water storage tank (RWST) into the RCS following a loss of coolant accident (LOCA). It is a low pressure, high capacity system.

The safety injection system is an emergency system located in the auxiliary building that also provides for injection of borated water from the RWST into the RCS in the event of a LOCA. It has a smaller capacity but a higher discharge pressure than the RHR system.

Attached to each reactor coolant loop's cold leg is a nitrogen-loaded accumulator, which will inject borated water into the RCS if the RCS pressure boundary ruptures (LOCA). When the pressure in the RCS drops below the pressure in the accumulators, the nitrogen will force

the borated water into the RCS to provide water to cover and cool the core and boron (a neutron absorber) to keep the core shutdown.

In the event a LOCA does occur, the hot water from the RCS will spill out into the containment and flash to steam, raising the pressure in the containment building. The containment spray pumps will transfer water from the RWST to spray rings located high inside containment. The cool water sprayed out into containment will quench the steam and return pressure inside containment to well within design limits. This limits the potential for the release of radioactive fission products in the coolant from the containment to the environment.

1.0.3 Plant Layout

The entire RCS (Figures 1.0-3 and 1.0-4), including the steam generators, is located in the containment building, which isolates the radioactive RCS from the environment in the event of a leak. The containment building is designed to contain the pressure produced by a complete rupture of an RCS loop. All potentially radioactive auxiliary systems are located in the auxiliary building, which is usually located between the turbine building and the containment. Systems which must be available to shutdown the reactor and/or mitigate the consequences of an accident are constructed to Seismic Category I standards, which means that they are designed to be capable of withstanding the maximum credible earthquake for the plant location. Buildings, such as the containment and auxiliary building, which house these systems and/or aid in minimizing any potential release of radioactivity to the environment, are also constructed to Seismic Category I requirements.

All ventilation from these buildings is passed through high efficiency particulate filters and/or charcoal filters to minimize radioactive releases. A fuel storage building (sometimes part of the auxiliary building) is provided for handling and storage of new and spent fuel. The fuel storage building is also a Seismic Category I building. The turbine building contains all of the secondary and secondary support systems. The main turbine and auxiliaries, moisture separator/reheaters, feedwater heaters, main condenser, condensate pumps, feedwater pumps, etc., are located in the turbine building. The turbine building is not a seismic structure.

1.0.4 Reactor Coolant System Pressure and Pressurizer Level Control

1.0.4.1 Pressure Control

The pressure in the RCS is maintained above the saturation pressure where bulk boiling could occur (a small amount of localized nucleate boiling is allowed). However, a system completely filled with water (solid) would be subject to very large pressure changes if the temperature of the fluid changes. To prevent this, the pressurizer is attached to one of the hot legs. It will act as a surge tank so expansion and contraction of the RCS water with temperature changes will not cause large pressure swings.

The pressurizer is maintained at saturation temperature for the desired RCS pressure (normally 2250 psia) by electrical heaters. This temperature ($\approx 653^{\circ}\text{F}$) is approximately 40°F hotter than the RCS hot leg temperature. Therefore, the only boiling that occurs in the RCS is in the pressurizer. The rest of the RCS is filled with water that is subcooled (temperature below that which will cause boiling to occur for the given pressure). Since it is maintained approximately half full of saturated water with the other half containing a steam volume, the pressurizer acts as a surge volume for the RCS.

If it is desired to increase RCS pressure, the heaters are energized to boil some of the water in the pressurizer and thus raise the pressure (the same mass of steam takes up about three times the space as the same mass of water). To reduce pressure, subcooled reactor coolant from the cold leg is sprayed into the steam volume to condense part of the steam bubble, which lowers the pressure. Since the pressurizer is directly connected to the RCS, these pressure changes are reflected in the entire system. Overpressure protection is provided by safety and relief valves connected to the pressurizer steam space.

1.0.4.2 Pressurizer Level Control

To optimize the pressure controlling abilities of the pressurizer, the correct steam/water volumes must be maintained. This is accomplished by maintaining a constant letdown flow (75 gpm) to the CVCS for cleanup, while varying the charging rate to raise or lower the water level in the pressurizer. For example, if the water level is lower than that required, the control system will raise the charging rate to some value above the 75 gpm letdown rate until the proper level is restored.

1.0.5 Reactor Control

There are three modes of control which may be used in a pressurized water reactor. All of these modes of control will be used to adjust reactor power in response to changes in certain measurable parameters, such as average reactor coolant system temperature (T_{avg}) or main steam header pressure (P_{stm}). The definition of average reactor coolant system temperature is as follows:

$$T_{avg} = \frac{RCS\ Hot\ Leg\ Temp. + RCS\ ColdLeg\ Temp.}{2}$$

The basic formula defining heat (or power) transferred across a heat exchanger (in our case the steam generators) is:

$$Q = UA\Delta T_{(SG)}$$

Where:

- Q = Heat transferred
- U = Heat transfer coefficient
- A = Area of heat transfer
- $\Delta T_{(SG)}$ = Differential temperature across the steam generator tubes, which is equal to the difference between T_{avg} and the main steam temperature (T_{stm}).

For all practical purposes, both the heat transfer coefficient (U) and the area of heat transfer (A) are assumed to be constant. The equation is then reduced to:

$$Q \propto \Delta T_{(SG)}$$

Or

$$Q \propto T_{avg} - T_{stm}$$

In order to increase power (Q), $\Delta T_{(SG)}$ must increase. The following describes plant control modes which can be used to control key plant parameters.

1.5.1 Constant T_{avg} Control Mode

With constant T_{avg} control (Figure 1.0-5), reactor power is adjusted to maintain a constant T_{avg} as turbine load is changed. Increasing turbine load causes T_{avg} to decrease as the turbine uses more energy than is produced in the reactor. The reactor control system senses this decrease in temperature and withdraws control rods to increase reactor power. An anticipatory signal comparing turbine and reactor power may also be utilized to optimize the transient response of the control system.

The constant T_{avg} control mode has the advantage of an unchanging RCS temperature and density, regardless of power level. Since the density does not change, pressurizer level is constant for all load conditions. This minimizes volume fluctuations in the RCS and reduces the use of the CVCS components in responding to the fluctuations.

A disadvantage of this control scheme is that it produces unacceptable secondary system steam conditions. In order for reactor power (Q) to increase with a constant T_{avg} , the saturation temperature (T_{stm}) in the secondary side of the steam generator must decrease as steam demand (turbine load) increases. This effect produces a decreasing main steam pressure (P_{stm}) with increasing turbine load. The low quality of the steam (high moisture content) that results at the last stages of the turbine may cause damage to the blading. This disadvantage far outweighs the advantage of constant pressurizer level. Therefore, constant T_{avg} control mode is not typically used in large PWRs.

1.0.5.2 Constant Steam Pressure Control Mode

With constant steam pressure control (Figure 1.0-6), reactor power is adjusted to maintain a constant steam pressure as turbine load is changed. As described in Section 1.0.5.1, increasing turbine load causes steam pressure (P_{stm}) to decrease. The reactor control system senses this decrease in P_{stm} and withdraws control rods to increase reactor power.

With this type of control system, the $\Delta T_{(SG)}$ is increased by raising T_{avg} and allowing P_{stm} and T_{stm} to remain constant. This produces excellent steam conditions for all loads from zero to 100% load. Disadvantages of this control scheme are that excessive rod motion is required and reactor hot leg temperature (T_h) can approach saturation values. Westinghouse designed PWRs do not utilize a constant steam pressure control mode.

1.0.5.3 Sliding T_{avg} Control Mode

A sliding T_{avg} control system (Figure 1.0-7) is a compromise between a constant T_{avg} and a constant steam pressure control. It retains the advantages of both but also retains some of the disadvantages. With a sliding T_{avg} control system, reactor power is adjusted to maintain a programmed increasing T_{avg} as turbine load is increased. As with the other control systems, increasing turbine load will cause T_{avg} and steam pressure to decrease. The control system will withdraw control rods to increase reactor power and maintain T_{avg} equal to program. An anticipatory circuit described in section 1.0.5.1 is also used in this type of control system. The $\Delta T_{(SG)}$ is increased by both raising T_{avg} and allowing T_{stm} (and thus P_{stm}) to decrease. This mode of control produces acceptable steam conditions at full load while requiring less rod motion and lower T_h than a constant steam pressure control. Most large PWRs utilize a sliding T_{avg} control system.

1.0.6 Plant Safety Limits

Safety limits for nuclear reactors are limits upon important process variables which are found to be necessary to reasonably protect the integrity of certain physical barriers which guard against the uncontrolled release of radioactivity.

1.0.6.1 Departure from Nucleate Boiling

One of the advantages of the sliding T_{avg} control mode over the constant steam pressure mode is the lower T_h that results. As stated in the section on constant steam pressure control, reactor outlet temperature (T_h) can approach saturation values. The problems associated with saturated conditions in the reactor coolant system will be discussed in this section.

Before describing the problems associated with saturated coolant, a review of the heat removal from the reactor core is in order. When the reactor is started up, the fission process generates heat in the fuel pellets. This heat is transferred from the fuel pellet through a helium fill gas to the fuel's zircaloy cladding. The flow of reactor coolant around the cladding transfers heat from the cladding. Reactor power is increased by increasing the number of

fissions taking place. An increase in fission rate increases fuel pellet temperature which in turn causes increased cladding temperatures. This increase in cladding temperature increases coolant temperature.

At some locations along the cladding surface, small steam bubbles will form. These bubbles form because the temperature of the cladding at these localized areas is hot enough to increase the temperature of the coolant to saturation and add enough energy to convert the water to steam. This localized formation of steam bubbles is called nucleate boiling (Figure 1.0-8).

Nucleate boiling increases the heat transfer from the cladding because of the agitating effect of localized bubble formation and collapse. Coolant flow sweeps the steam bubbles from the cladding and relatively colder water replaces the bubbles. The steam bubbles transfer their energy to the coolant because the coolant temperature is less than the steam bubble temperature. Nucleate boiling is a very important heat transfer mechanism in pressurized water reactors at high power levels.

As heat generation within the fuel increases, the rate of bubble formation on the cladding increases. As a result, the bubbles occupy a greater percentage of the cladding surface area. Further increases in heat generation will increase steam bubble formation to a point where they are being produced faster than they can be swept away by coolant flow. Eventually, the fuel cladding will be covered by steam bubbles, and direct contact between the coolant and cladding is prevented. This layer of steam bubbles now serves as an “insulation” impeding heat transfer from the fuel and cladding. This condition is known as partial film boiling.

Partial film boiling is not permitted because the insulating effect of the steam bubbles causes rapid increases in cladding temperature that can lead to cladding failures. Since the cladding functions to prevent the escape of fission products, failure causes a release of fission products to the coolant. It would be quite simple to prevent cladding failures by not allowing any boiling to occur. However, the advantages of the high heat transfer from nucleate boiling would be lost. The problem now becomes one of allowing nucleate boiling and its associated benefits while preventing the detrimental effects of partial film boiling. In other words, the departure from nucleate boiling (DNB) must be prevented.

The problem can be solved if the level of heat energy (heat flux) in the cladding can be maintained below a value that will prevent the transition from nucleate boiling to partial film boiling. If this level represents the departure from nucleate boiling, then the ratio of the heat energy required for departure from nucleate boiling to the actual local heat flux at a given reactor power level will represent the approach to potential cladding damage. This ratio is known as the departure from nucleate boiling ratio (DNBR), and for the purposes of this discussion will be defined as:

$$DNBR = \frac{\text{heat flux required for DNB to occur}}{\text{actual local heat flux}}$$

The reactor is assumed to be operating in or below the nucleate boiling heat transfer region if DNBR is greater than one. At DNBR values less than one, partial film boiling is assumed to occur. At DNBR equal to one, great difficulty exists in determining exactly what will happen. Therefore, a value greater than one has been conservatively chosen as the DNBR limit. For Westinghouse designed PWRs, the minimum DNBR allowed is 1.3. Since maintaining DNBR within acceptable limits is necessary for cladding integrity, it is designated as one of the plant's safety limits.

Thus far in this discussion, the effect of power level (heat flux) and its influence on DNBR has been described. However, reactor coolant system pressure, temperature, flow, and the distribution of power also affect DNBR. If reactor coolant system pressure is decreased, the DNBR will decrease because less heat flux (lower power) is required to cause film boiling. Conversely, an increase in pressure will increase DNBR, and a higher heat energy is allowed. In summary, DNBR is directly proportional to pressure.

The effect on DNBR of operating at a higher reactor coolant system temperature can be explained if one considers that the higher temperature represents a higher heat energy in the coolant. Therefore, less heat energy is required to cause film boiling to occur. DNBR is inversely proportional to reactor coolant system temperature.

Reactor coolant system flow facilitates heat removal from the cladding. A decrease in RCS flow results in a decrease in heat removal capability. A decrease in heat removal reduces DNBR.

Modern PWR cores are about 12 feet high, and the distribution of power (heat flux) in the core has an important effect on DNBR. For example, if a greater portion of the total power is being produced in the bottom half of the core, the "cold" inlet coolant provides sufficient heat removal to prevent DNBR problems. However, if a greater portion of the total power is being produced in the top half of the core, the DNBR will decrease due to the decrease in heat removal from the higher temperature coolant. In general, top peaked power distributions are worse from a DNBR standpoint.

To ensure that the DNBR will remain at acceptable values, the combined effects of total power, RCS pressure, RCS temperature, RCS flow, and power distribution are monitored by the reactor protection system to automatically shutdown (trip) the reactor if the limiting value of DNBR is approached. RCS differential temperature (ΔT) is used by the reactor protection system as a diverse indication of power level for the DNBR and kW/ft reactor trips. RCS differential temperature is defined as (NOTE: RCS ΔT is not the same as $\Delta T_{(SG)}$ discussed in section 1.0.5):

$$\Delta T = \text{RCS hot leg temp.} - \text{RCS cold leg temp.}$$

1.0.6.2 Power Density (kW/ft)

The second plant safety limit also deals with heat energy production, and like DNBR, ensures that the cladding barrier remains intact. This limit is a power density limit imposed to prevent centerline fuel temperature from exceeding the melting temperature. Since fuel temperature is not directly measurable, the limit is expressed in heat energy production per foot of fuel rod (kW/ft).

The melting temperature of the uranium oxide fuel is approximately 5000°F. If this temperature is approached, the thermal expansion of the fuel pellets can cause excessive cladding stresses. Power density is directly proportional to total power and power distribution. To ensure that cladding integrity is maintained, the reactor protection system will automatically trip the reactor before kW/ft limits are exceeded.

1.0.6.3 Reactor Coolant System Pressure

RCS pressure is the last safety limit imposed to ensure the integrity of the second barrier to the release of fission products. The RCS has a design pressure of 2500 psia. The safety limit is 110% of design pressure, or 2750 psia, and is maintained by the code safety valves on the pressurizer and a high pressurizer pressure reactor trip generated by the reactor protection system.

1.0.7 Summary

PWRs use a dual cycle concepts where the closed primary cycle is separate from the steam cycle. The point of heat transfer between the two cycles is the steam generators. The RCS is the primary cycle and is located in the containment building. The secondary cycle is the steam system, the turbine-generator (where steam energy is used to generate electrical power), and the condensate and feedwater systems.

The secondary cycle equipment and systems are principally located in the turbine building. Support and emergency systems provide several functions to both the primary and secondary cycles.

Systems, components, and buildings which have safety functions or are required to help maintain the integrity of the RCS or core are built to Seismic Category I standards and are capable of withstanding the maximum credible seismic event.

Reactor control modes such as “constant T_{avg} ” mode and “constant steam pressure” mode can be used, but large Westinghouse units use the “sliding T_{avg} ” mode. The “sliding T_{avg} ” mode is a compromise between the other two and contains some of the advantages and disadvantages of both.

There are three safety limits for Westinghouse PWRs. These are:

- Departure from nucleate boiling ratio,
- Power density or kW/ft, and
- Reactor coolant system pressure.

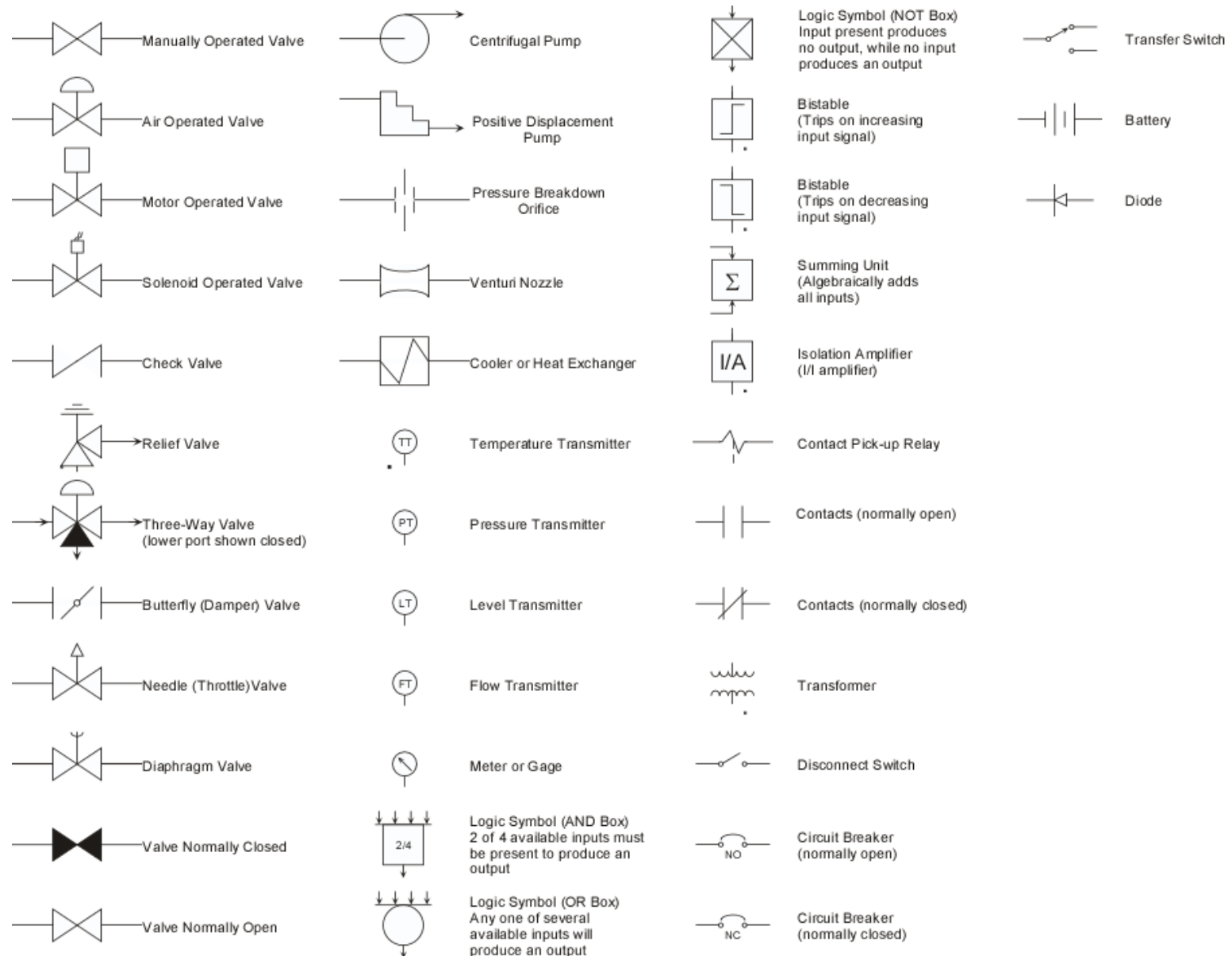


Figure 1.0-1, List of Symbols

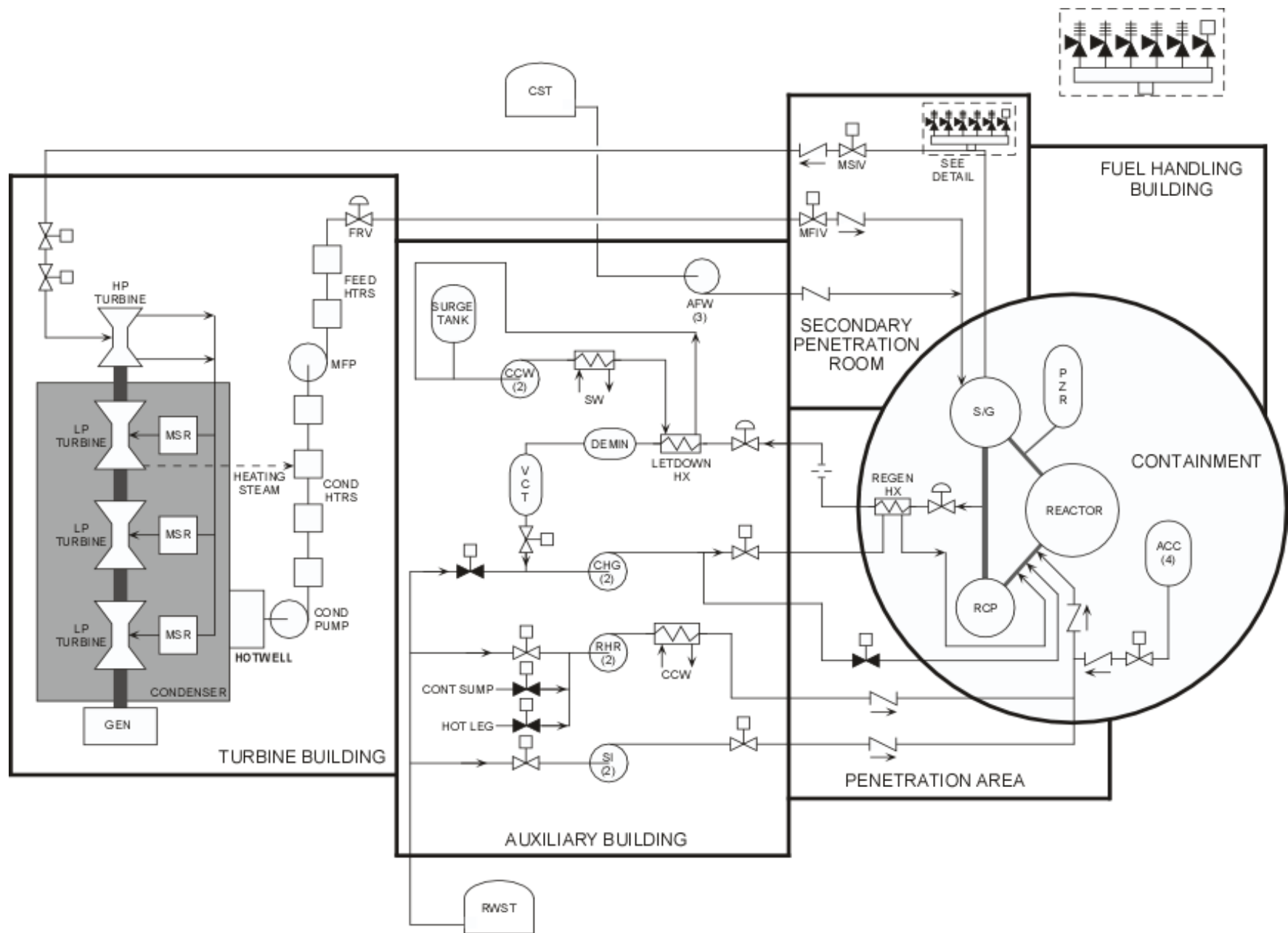


Figure 1.0-2, Plant Systems Composite

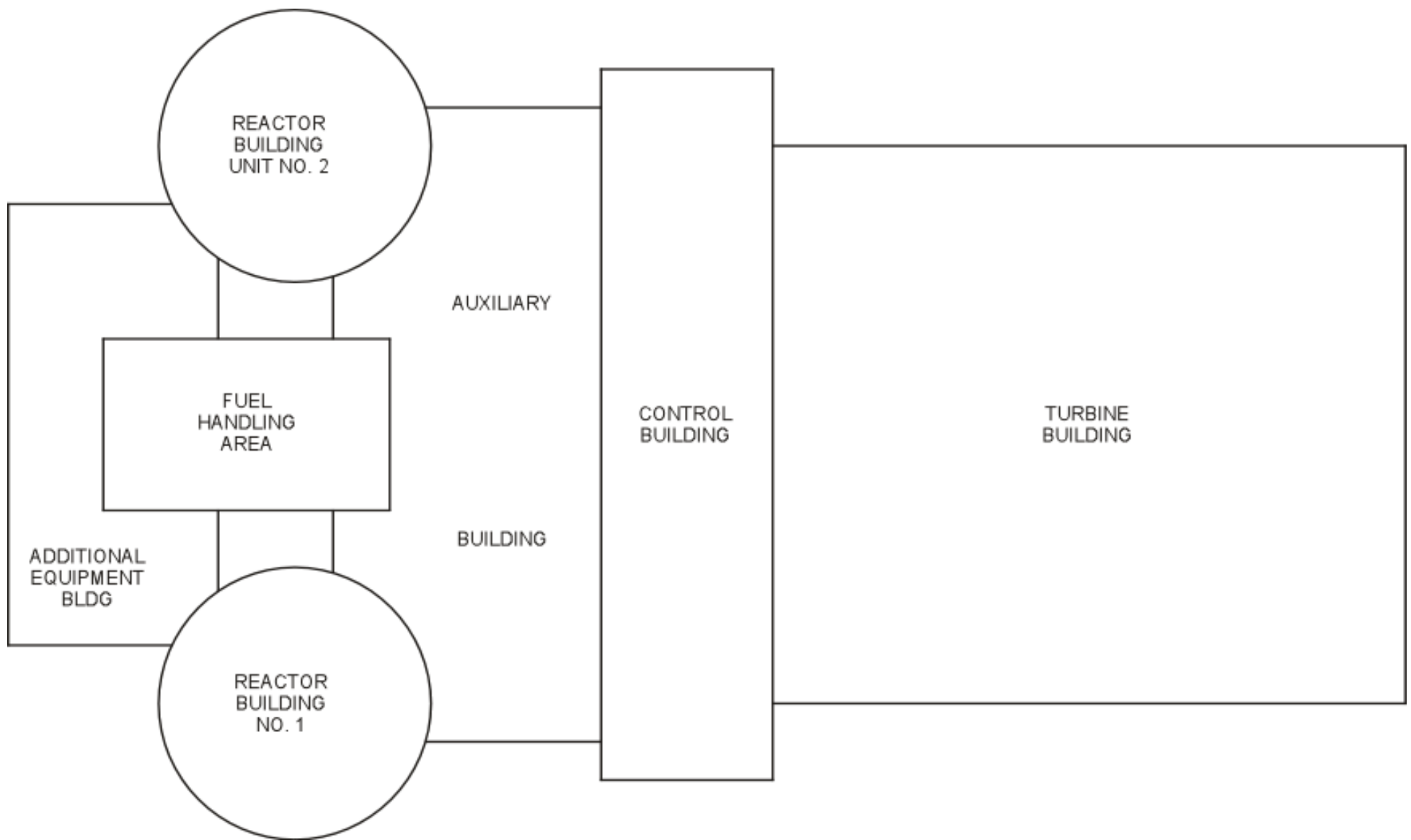


Figure 1.0-3, Plant Layout

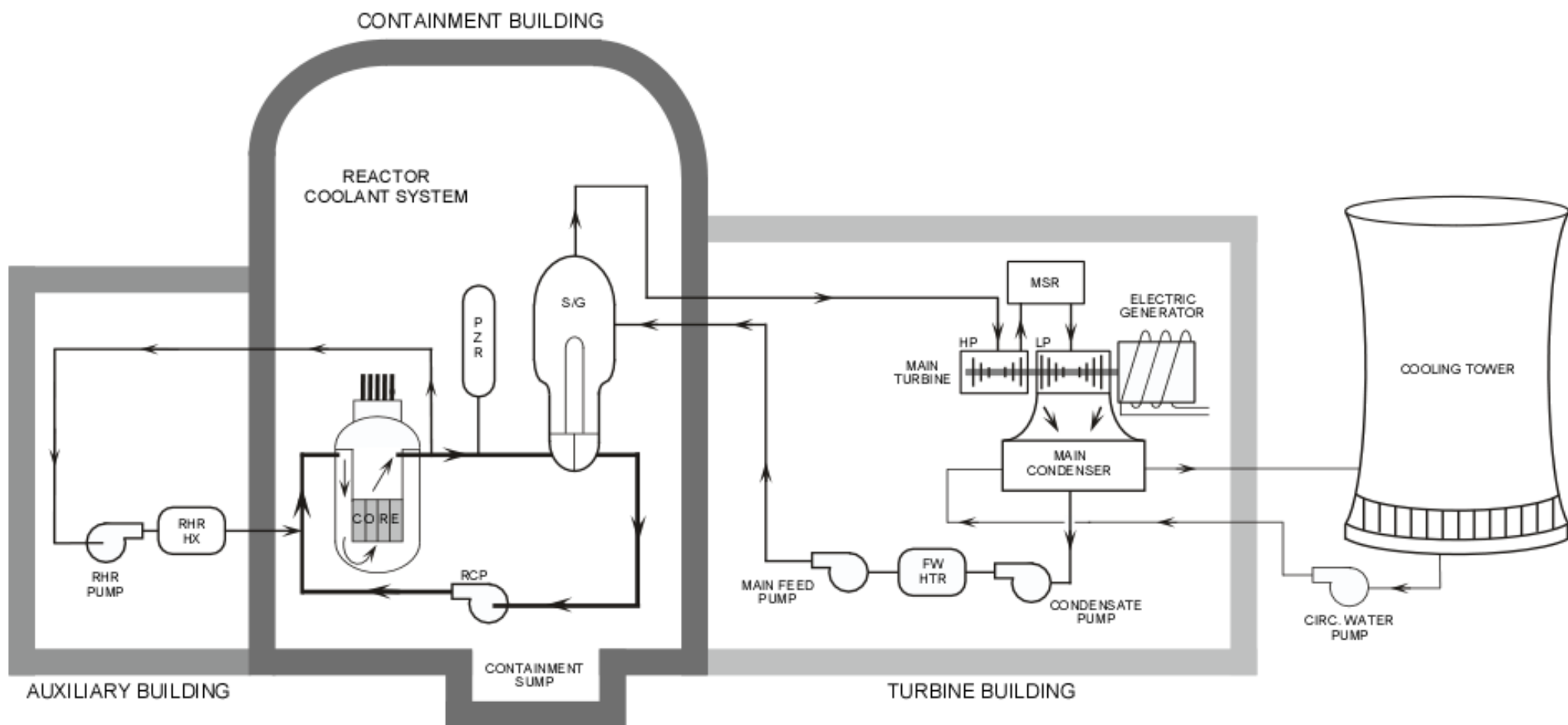


Figure 1.0-4, Basic PWR Arrangement

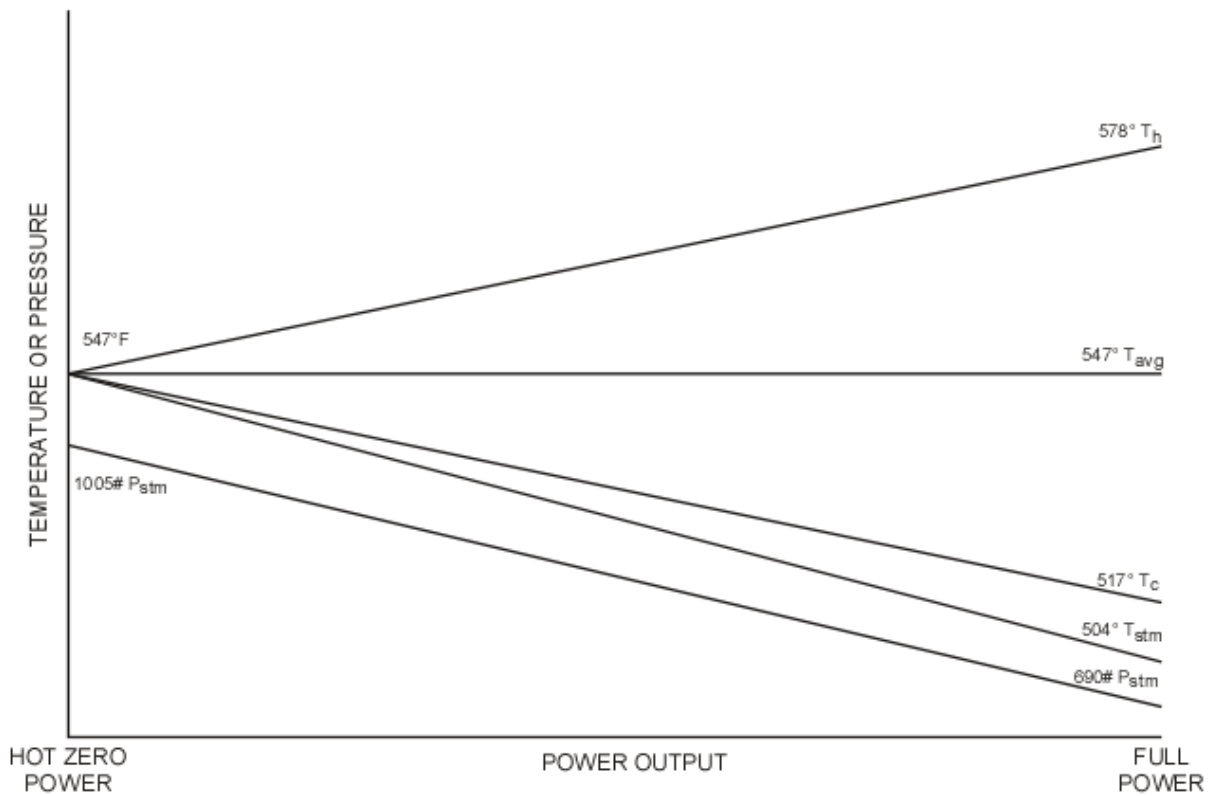
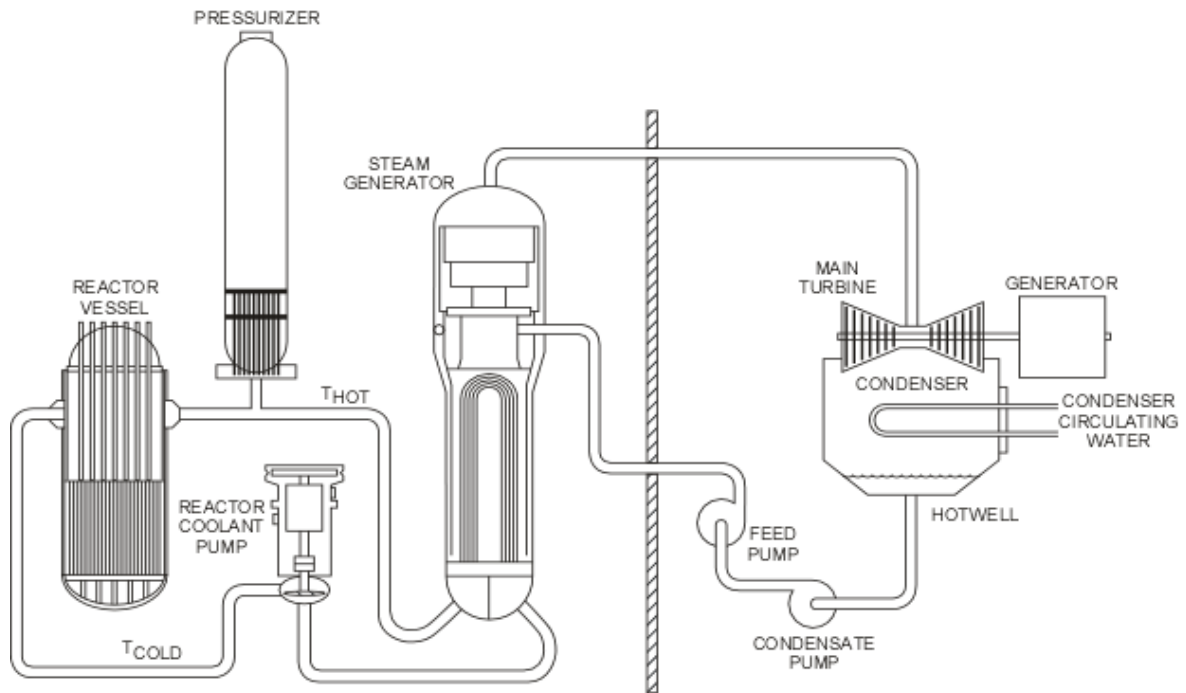


Figure 1.0-5, Characteristics of a Constant Average Temperature Program

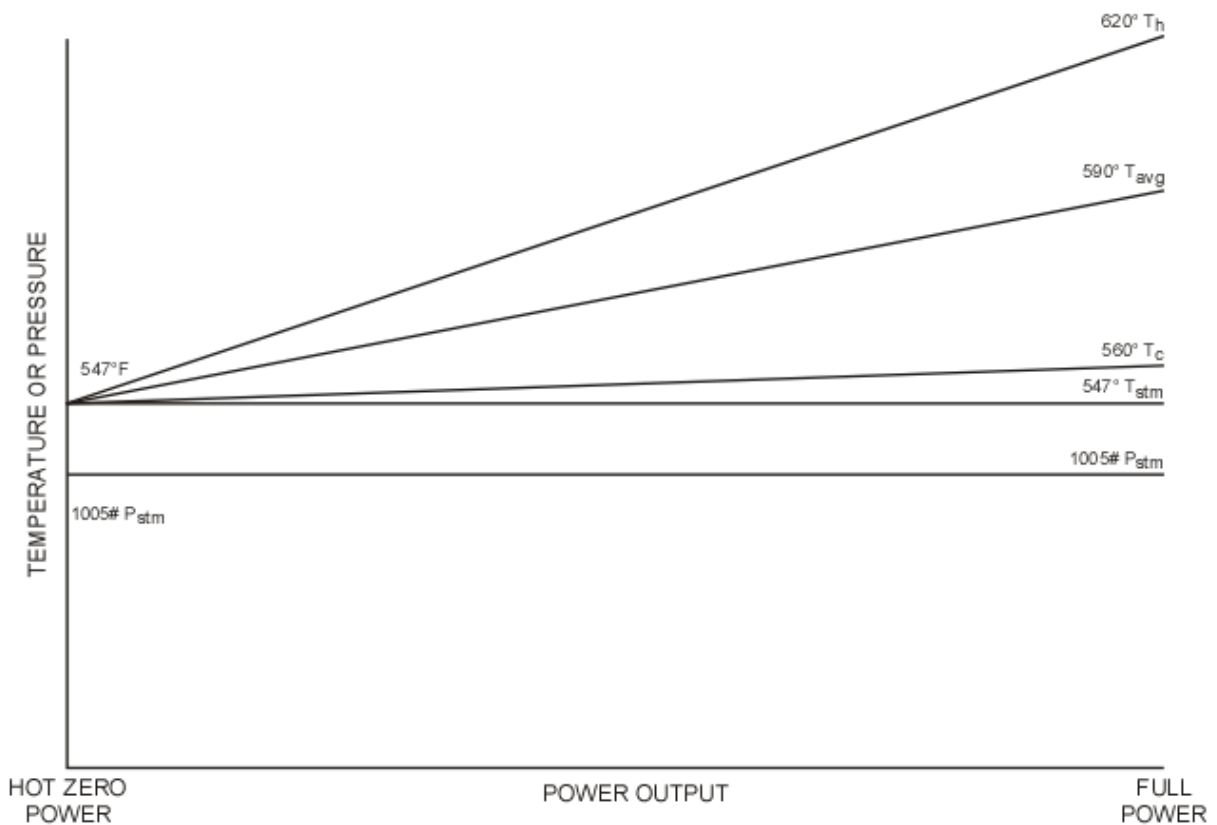
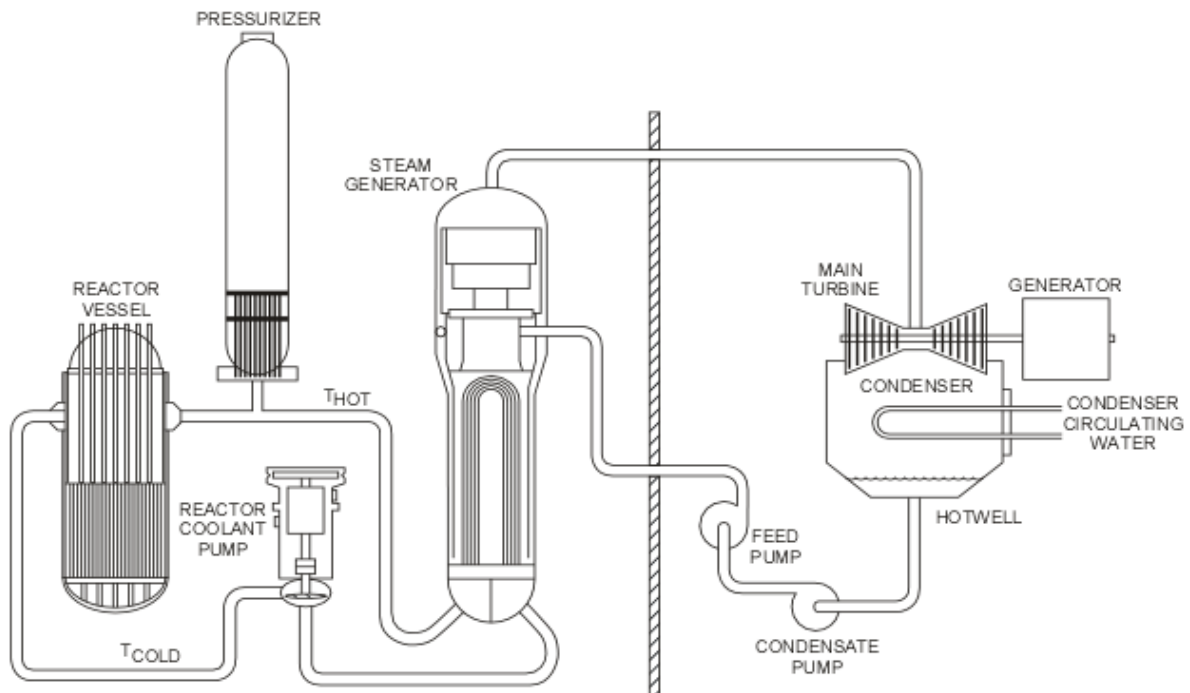


Figure 1.0-6, Characteristics of a Constant Steam Pressure Program

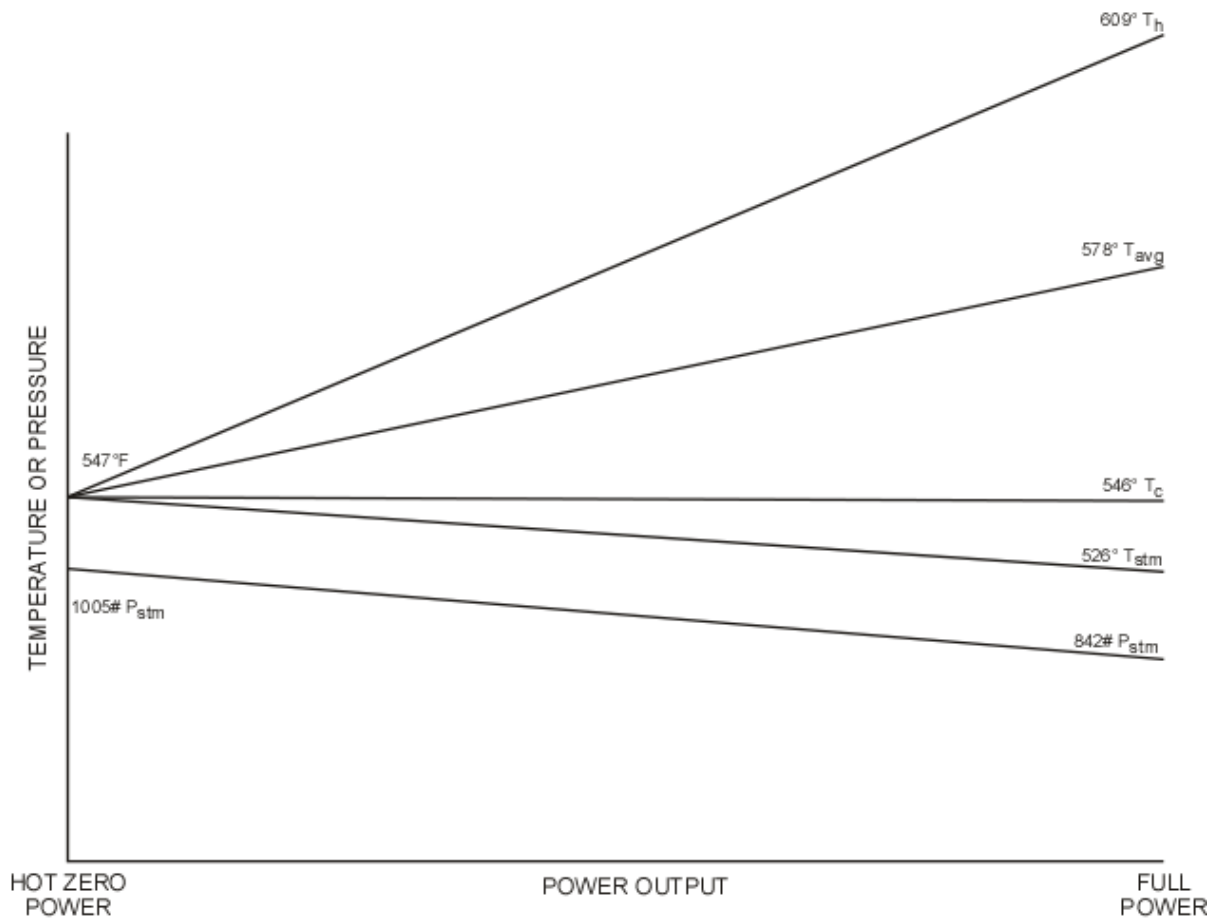
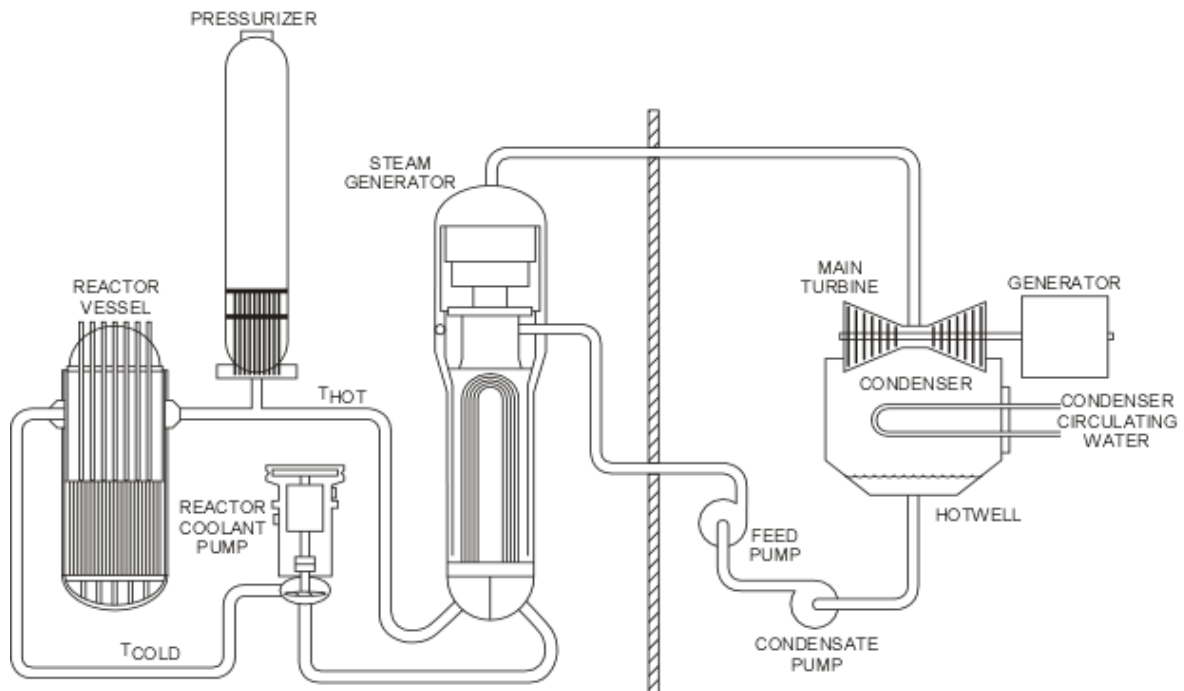
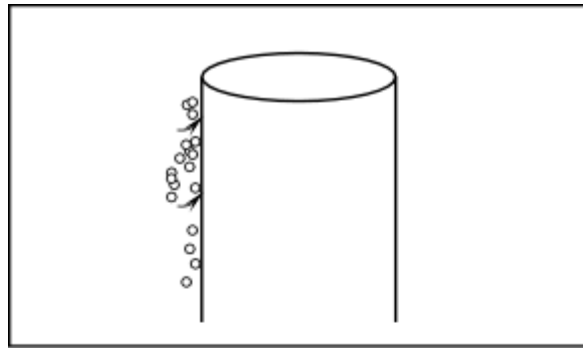
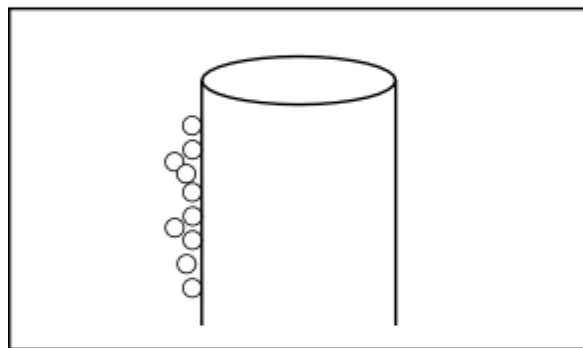


Figure 1.0-7, Characteristics of a Sliding Average Temperature Program



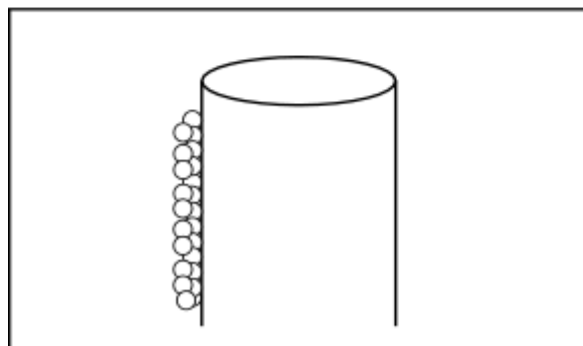
CONVECTION CURRENTS
ENHANCE COOLING AT
THE CLAD SURFACE

NUCLEATE BOILING



RADIATIVE HEAT TRANSFER
ACROSS LARGER STEAM
BUBBLES ON THE CLAD SURFACE
-LESS EFFICIENT-

PARTIAL FILM BOILING = DNB



STEAM BLANKETING ON THE
CLAD SURFACE - RADIATIVE
AND
CONDUCTIVE HEAT TRANSFER
-LEAST EFFICIENT-

FILM BOILING

Figure 1.0-8, Departure From Nucleate Boiling